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DEC 19 2012

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 312 TO UNIT 1 LICENSE NPF-14
AND PROPOSED AMENDMENT NO. 284 TO UNIT 2 LICENSE NPF-22:
LOW PRESSURE SAFETY LIMIT AND REFERENCE
CHANGES **Docket Nos. 50-387**
PLA-6915 **and 50-388**

In accordance with the provisions of 10 CFR 50.90, PPL Susquehanna, LLC (PPL) is submitting a request for an amendment to the Technical Specifications (TS) for Susquehanna Steam Electric Station (SSES) Units 1 and 2.

The proposed amendment would modify TS Section 2.1.1 to reflect a revised Low Pressure Safety Limit. The change to the limit became necessary as a result of GE PART 21 REPORT, SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit."

The proposed changes have been reviewed by both the Plant Operations Review Committee (PORC) and the Susquehanna Review Committee (SRC).

The enclosure to this letter contains PPL's evaluation of the proposed change. Included are a description of the proposed change, technical analysis of the change, regulatory analysis of the change, and the environmental considerations associated with the change. The enclosure also contains the following attachments:

- Attachment 1 provides the existing Technical Specifications pages marked-up to show the proposed changes.
- Attachment 2 provides the existing Bases pages marked-up to show the proposed changes.

There are no new regulatory commitments contained in this letter.

PPL requests NRC complete its review of this change by December 31, 2013 with the amendments being implemented within 60 days following approval.

In accordance with 10 CFR 50.91(b), a copy of this application, with its attachments, is being provided to the designated Commonwealth of Pennsylvania state official.

If you should have any questions regarding this submittal, please contact Mr. John Tripoli at (570) 542-3100.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 12/19/12

J. Helsel

Enclosure: PPL Evaluation of the Proposed Changes, Unit 1 and Unit 2 Low Pressure Safety Limit

Attachments:
Attachment 1 - Proposed Technical Specification Changes, (Mark-ups)
Attachment 2 - Proposed Technical Specification Bases Changes, (Mark-ups Provided for Information)

Copy: NRC Region I
Mr. P. W. Finney, NRC Sr. Resident Inspector
Mr. J. Whited, NRC Project Manager
Mr. L. J. Winker, DEP/BRP

ENCLOSURE to PLA-6915

PPL Evaluation of Proposed Changes

Unit 1 and Unit 2 Low Pressure Safety Limit

ATTACHMENTS:

Attachment 1 – Proposed Technical Specification Changes, (Mark-ups)

Attachment 2 – Proposed Technical Specification Bases Changes, (Mark-ups Provided for Information)

PPL EVALUATION OF PROPOSED CHANGES

1.0 SUMMARY DESCRIPTION

This letter is a request to amend Operating License NPF-14 for Susquehanna Steam Electric Station (SSES) Unit 1 and Operating License NPF-22 for SSES Unit 2. Specifically, the proposed changes would modify the SSES Unit 1 and SSES Unit 2 Technical Specifications (TS) Section 2.1.1 to reflect a revised Low Pressure Safety Limit. The change to TS Section 2.1.1 became necessary as a result of General Electric (GE) PART 21 REPORT, SC05-03, Potential to Exceed Low Pressure Technical Specification Safety Limit.

The proposed changes are described in detail in Section 2.0.

2.0 DETAILED DESCRIPTION

2.1 Proposed Changes

The proposed changes would revise the reactor steam dome pressure value in TS 2.1.1.1 and 2.1.1.2 from 785 psig to 557 psig. This change is required to reflect that the SPCB correlation (Reference 6.3) is valid for critical power calculations at pressures $\geq$ 571.4 psia. The associated TS Bases changes corresponding to the proposed TS change are included for information.

2.2 Background

Initially the Boiling Water Reactor Owners' Group (BWROG) attempted resolve the Part 21 issue; however, in April 2012 the BWROG discontinued the effort and recommended that plants lower their Low Pressure Safety Limit to meet the lower range of their critical power correlation.

Excessive thermal overheating of the fuel rod cladding can result in cladding damage and the release of fission products. In order to protect the cladding against thermal overheating due to boiling transition, the Safety Limits in Section 2.1.1 of the SSES Unit 1 and Unit 2 TS were established.

General Design Criterion (GDC) 10 requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). TS 2.1.1 ensures compliance with GDC 10 by setting reactor conditions such that no significant fuel damage will occur if the

conditions are met. The reactor conditions currently specified in TS 2.1.1.1 are "With the reactor steam dome pressure < 785 psig or core flow < 10 million lbm/hr: THERMAL POWER shall be ≤ 23% RTP." The 785 psig value was based on the lower value for the applicability of the GE MCPR Methodology (the GEXL correlation) at the time the plant was licensed, and ensured a valid CPR calculation was performed for the AOOs described in the Final Safety Analysis Report (FSAR).

GE Part 21 Notification SC05-03 identifies an AOO, the Pressure Regulator Failure Open (PRFO) event that could potentially violate the requirements of TS 2.1.1.1. Specifically, during the PRFO event, reactor steam dome pressure could be < 785 psig with reactor thermal power > 23%. Susquehanna TS 2.1.1.1 requires that core thermal power be ≤ 23%, when reactor steam dome pressure is < 785 psig or core flow is < 10 million lbm/hr. In addition, the TS Bases for the Main Steam Line Pressure - Low trip setpoint (TSB 3.3.6.1) states that the "...Function is directly assumed in the analysis of the pressure regulator failure" and that "... this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded."

The proposed change to TS 2.1.1.1 continues to ensure that a valid CPR calculation is performed for AOOs described in the FSAR including the PRFO.

The reactor conditions currently specified in TS 2.1.1.2 are "With the reactor steam dome pressure < 785 psig or core flow < 10 million lbm/hr: MCPR shall be ≥1.09 for two recirculation loop operation or ≥1.12 for single recirculation loop operation." The proposed change to TS 2.1.1.2 is consistent with the proposed change to TS 2.1.1.1.

3.0 TECHNICAL EVALUATION

The changes described in Section 2.0 were made necessary by GE Part 21 Notification SC05-03, and were made possible because the current NRC approved MCPR methodology for SSES, the SPCB correlation, (Reference 6.3) is valid for critical power calculations at pressures ≥571.4 psia. The 785 psig Safety Limit in TS 2.1.1.1 originates from a limitation on the GE CPR correlation (GEXL) that was used at the time of plant licensing. The initial GEXL was only licensed to 785 psig.

GDC 10 requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). TS 2.1.1.1 ensures compliance with GDC 10 by setting reactor conditions such that no significant fuel damage will occur if the conditions are met.

The Pressure Regulator Failure Open (PRFO) is an AOO and is described in FSAR Section 15.1.3. During the PRFO, the pressure regulation system fails such that a maximum demand signal is issued. The signal results in the Turbine Control Valves (TCVs) and Bypass valves (BPVs) opening to the position allowed by the Maximum Combined Flow Limiter (MCFL). This results in a rapid reactor vessel depressurization. The event can be terminated by one of three automatic signals:

1) High water level due to the level swell from the voiding in the core,
2) High Main Steam Line Flow, and
3) Low Main Steam Line Pressure.

GE Part 21 Notification SC05-03 states that the PRFO event may violate the requirements of TS 2.1.1.1. The PRFO is non-limiting for fuel cladding integrity because the Critical Power Ration (CPR) increases during the event.

GE analyzed the PRFO as part of the original licensing of each BWR. The results of the analysis showed that the event would terminate on high water level before the requirements of TS 2.1.1.1 were violated. GE reanalyzed the PRFO using improved methods that show the event may instead be terminated by low main steam line pressure. Therefore, depending on a variety of inputs the reactor pressure during the PRFO may decrease below 785 psig before reactor thermal power is less than 23%, and hence violate TS 2.1.1.1. The inputs that affect the PRFO response include: the Main Steam Line Pressure - Low setpoint (TS 3.3.6.1), MCFL setting, steam line pressure drop, initial reactor power, and utility operating strategies (e.g., final feedwater temperature reduction).

The 785 psig Safety Limit in TS 2.1.1.1 originates from a limitation on the GE CPR correlation, GEXL, used at the time the plant was licensed. The initial GEXL was only licensed to 785 psig. Therefore, CPR calculations by GEXL below 785 psig were not valid to determine CPR margin for normal operation or AOOs.

PPL uses AREVA's NRC approved critical power correlation, SPCB, for MCPR Safety Limit determination, reload licensing analyses, and MCPR monitoring. SPCB is currently included in TS 5.6.5.b. Per TSB 2.1.1.1, SPCB is approved for CPR calculations by the NRC for reactor pressures > 556.7 psig (571.4 psia - 14.7 psia). Since the intent of TS 2.1.1.1 is to prevent the operation in a region where the CPR calculation is invalid, the reactor pressure in TS 2.1.1.1 may be lowered to 557 psig.

PPL has determined that the PRFO will not violate the proposed Low Pressure Safety Limit of 557 psig.

It should be noted that the nominal and allowable trip setpoints for the Main Steam Line Pressure - Low are 861 psig (Item 2.2.2.3.2 from TRM Table 2.2-1) and 841 psig (Item 1.b from TS Table 3.3.6.1-1), respectively. These setpoints provide added assurance that the revised criteria of 557 psig from TS 2.1.1.1 would not be violated under realistic conditions.

In summary, based on the above, it is expected that the PRFO event may violate the current safety limit of TS 2.1.1.1. As a result, PPL intends to change the value to 557 psig. The SPCB CPR correlation is licensed to pressures > 557 psig and would show acceptable CPR values with the current Main Steam Line Pressure - Low setpoint.

4.0 REGULATORY SAFETY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

Title 10 of the Code of Federal Regulations (10 CFR) establishes the fundamental regulatory requirements with respect to reactivity control systems. Specifically, General Design Criterion 10 (GDC 10), " Reactor design," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 states, in part, that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded.

As long as the core pressure and flow are within the range of validity of the SPCB correlation, the proposed Low Pressure Safety Limit values in TS Section 2.1.1 will ensure that 99.9% of the fuel rods in the core are not expected to experience boiling transition. This satisfies the requirements of GDC 10 regarding acceptable fuel design limits.

4.2 No Significant Hazards Consideration

PPL has evaluated the proposed changes using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration. An analysis of the issue of no significant hazards consideration is presented below.

(1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment changes the low pressure safety limit in Technical Specification (TS) 2.1.1 from 785 psig to 557 psig based on the capabilities of the current critical power correlation used by Susquehanna (SPCB). The SPCB correlation is approved for CPR calculations by the NRC for reactor pressures > 571.4 psia and is listed as an approved analytical method in TS 5.6.5.b.

The proposed changes will not alter existing Final Safety Analysis Report (FSAR) design basis accident analysis assumptions, add any accident initiators, or affect the function of the plant safety-related structures, systems, or components (SSCs) as to how they are operated, maintained, modified, tested, or inspected. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The change to the Low Pressure Safety Limits does not result in the need for any new or different FSAR design basis accident analysis. The inclusion does not introduce

new equipment that could create a new or different kind of accident, and no new equipment failure modes are created. In addition, the proposed change does not affect the function of any safety-related SSC as to how they are operated, maintained, modified, tested or inspected. As a result, no new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed amendment. Therefore, the proposed amendment does not create a possibility for an accident of a new or different type than those previously evaluated.

3. ***Does the proposed amendment involve a significant reduction in a margin of safety?***

   Response: No.

   The margin of safety is associated with the confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation to the public. Evaluation of the 10 CFR 21 issue that identified the need for the proposed change determined that there was no decrease in the safety margin and therefore no threat to fuel cladding integrity. The proposed changes to the Low Pressure Safety Limits would not alter the way safety-related SSCs function and would not alter the way PPL Susquehanna Units 1 and 2 are operated. The proposed changes to the safety limit are within the capabilities of the existing NRC approved CPR correlation and ensure valid CPR calculations for the Anticipated Operational Occurrences (AOOs) defined in the FSAR. The proposed amendment would have no impact on the structural integrity of the fuel cladding, reactor coolant pressure boundary, or containment structure. Based on the above considerations, the proposed amendment would not degrade the confidence in the ability of the fission product barriers to limit the level of radiation to the public. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above, PPL Susquehanna, LLC (PPL) concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. PPL Susquehanna, LLC has evaluated the proposed changes and has determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

As demonstrated in the No Significant Hazards Consideration Evaluation, the proposed amendment does not involve a significant hazards consideration.

There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

There is no significant increase in individual or cumulative occupational radiation exposure. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

6.0 REFERENCES

6.1. SC05-03, 10 CFR Part 21 Communication. "Potential to Exceed Low Pressure Technical Specification Safety Limit," March 29, 2005.

6.2. NRC MFN 05-021, "10CFR21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," March 29, 2005.

6.3. EMF-2209(P)(A), Revision 3, "SPCB Critical Power Correlation," AREVA NP, September 2009.

Attachment 1 to PLA-6915

Proposed Technical Specification Changes (Mark-ups)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~ 557 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 557 psig and core flow ≥ 10 million lbm/hr:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~ 557 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 557 psig and core flow ≥ 10 million lbm/hr:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Attachment 2 to PLA-6915

Proposed Technical Specification Bases Changes (Mark-ups Provided for Information)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 for AREVA NP~~Siemens Power Corporation~~ fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

(continued)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

The use of the SPCB (Reference 4) correlation is valid for critical power calculations at pressures ≥ 571.4 psia and bundle mass fluxes > 0.087×10^6 lb/hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition.

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BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

For the ~~FANP~~ AREVA NP ATRIUM-10 design, the minimum bundle flow is > 28 x 10^3 lb/hr. For the AREVA NP ATRIUM-10 fuel design, the coolant minimum bundle flow and maximum area are such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power test data taken from various ~~SPC~~ AREVA NP and GE fuel designs at pressures from 14.7 psia to 1400 psia indicate the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 23% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 2.8, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 23% RTP for reactor pressures < ~~785~~ 557 psig is conservative and for conditions of lesser power would remain conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty in the critical power correlation. References 2, 4, and 5 describe the methodology used in determining the MCPR SL.

The SPCB critical power correlation is based on a significant body of practical test data. As long as the core pressure and flow are within the range of validity of the correlations (refer to Section B.2.1.1.1), the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the SPCB correlation provide a reasonable degree of assurance that during sustained operation at the MCPR SL there would be no transition boiling in the core.

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APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised.

Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

~~SPC Atrium~~ AREVA NP ATRIUM -10 fuel is monitored using the SPCB Critical Power Correlation. The effects of channel bow on MCPR are explicitly included in the calculation of the MCPR SL. Explicit treatment of channel bow in the MCPR SL addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

Monitoring required for compliance with the MCPR SL is specified in LCO 3.2.2, Minimum Critical Power Ratio.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be

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BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of regulatory limits. Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ANF-524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.

3. Deleted.

4. EMF-2209(P)(A), "SPCB Critical Power Correlation," ~~Framatome ANP~~AREVA NP, [See Core Operating Limits Report for Revision Level].

5. EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2," October 1999.

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 for AREVA NP ~~Siemens Power Corporation~~ fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

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BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

The use of the SPCB (Reference 4) correlation is valid for critical power calculations at pressures ≥ 571.4 psia and bundle mass fluxes > 0.087 x 10^6 lb/hr-ft^2 for SPCB. For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the ~~FANP Atrium~~ AREVA NP ATRIUM-10 design, the minimum bundle flow is > 28 x 10^3 lb/hr. For ~~Atrium~~ AREVA NP ATRIUM-10 fuel design, the coolant minimum bundle flow and maximum area are such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power test data taken from various ~~SPC~~ AREVA NP and GE fuel designs at pressures from 14.7 psia to 1400 psia indicate the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 23% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 2.8, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 23% RTP for reactor pressures < ~~785~~ 557 psig is conservative and for conditions of lesser power would remain the same.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure

(continued)

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty in the critical power correlation. References 2, 4 and 5 describe the methodology used in determining the MCPR SL.

The SPCB critical power correlation is based on a significant body of practical test data. As long as the core pressure and flow are within the range of validity of the correlation (refer to Section B 2.1.1.1), the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the SPCB correlation provide a reasonable degree of assurance that during sustained operation at the MCPR SL there would be no transition boiling in the core. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised.

Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

~~SPC~~ AREVA NP ATRIUM-10 fuel is monitored using the SPCB Critical Power Correlation. The effects of channel bow on MCPR are explicitly included in the calculation of the MCPR SL. Explicit treatment of channel bow in the MCPR SL addresses the concerns of the NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

Monitoring required for compliance with the MCPR SL is specified in LCO 3.2.2, Minimum Critical Power Ratio.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of regulatory limits. Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ANF~~B~~-524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.

3. Deleted.

4. EMF-2209(P)(A), ~~Revision 2,~~ "SPCB Critical Power Correlation," AREVA NP, [See Core Operating Limits Report for Revision Level]. ~~Siemens Power Corporation, September 2003~~.

5. EMF-2158(P)(A), Rev. 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2," October 1999.

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